B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

AEC GDC Criterion 6 (Ref. 1) requires that the reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. This integrity is required during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient.

BASES

BACKGROUND (continued)

Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System allowable values specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation", in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude DNB related flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Overtemperature ΔT trip;

d. Overpower ΔT trip;

e. Power Range Neutron Flux trip; and

f. Steam generator safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS allowable values identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow-Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the USAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation.

The SL curves in Figure 2.1.1-1 define the regions of acceptable operation with respect to average temperatures, power (measured in ΔT), and pressurizer pressure. Each of the curves in the Figure has three slopes. For the 2235 and 2385 psig curves, at lower power (lower ΔT) the vessel exit design temperature, 650°F, is limiting.

BASES

SAFETY LIMITS (continued)

For the lower pressure curves, at lower ΔT, vessel exit temperature T_{sat} is limiting, to ensure the ΔT measurement remains valid. At all pressures after the first knee, at higher ΔT, the minimum DNBR derived from the critical heat flux correlation is limiting. The change in slope near full power ΔT is due to more restrictive $F_{\Delta H}$ consideration in the DNBR limit at high power.

The curves are based on enthalpy hot channel factor limits provided in the CORE OPERATING LIMITS REPORT (COLR). Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.

The SL is higher than the setpoint calculated when the AXIAL FLUX DIFFERENCE (AFD) is within the limits of the $f(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 3).

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Allowable values for the reactor trip functions are specified in LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.” In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 6, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 14.3.

3. WCAP-13123, December 1991.

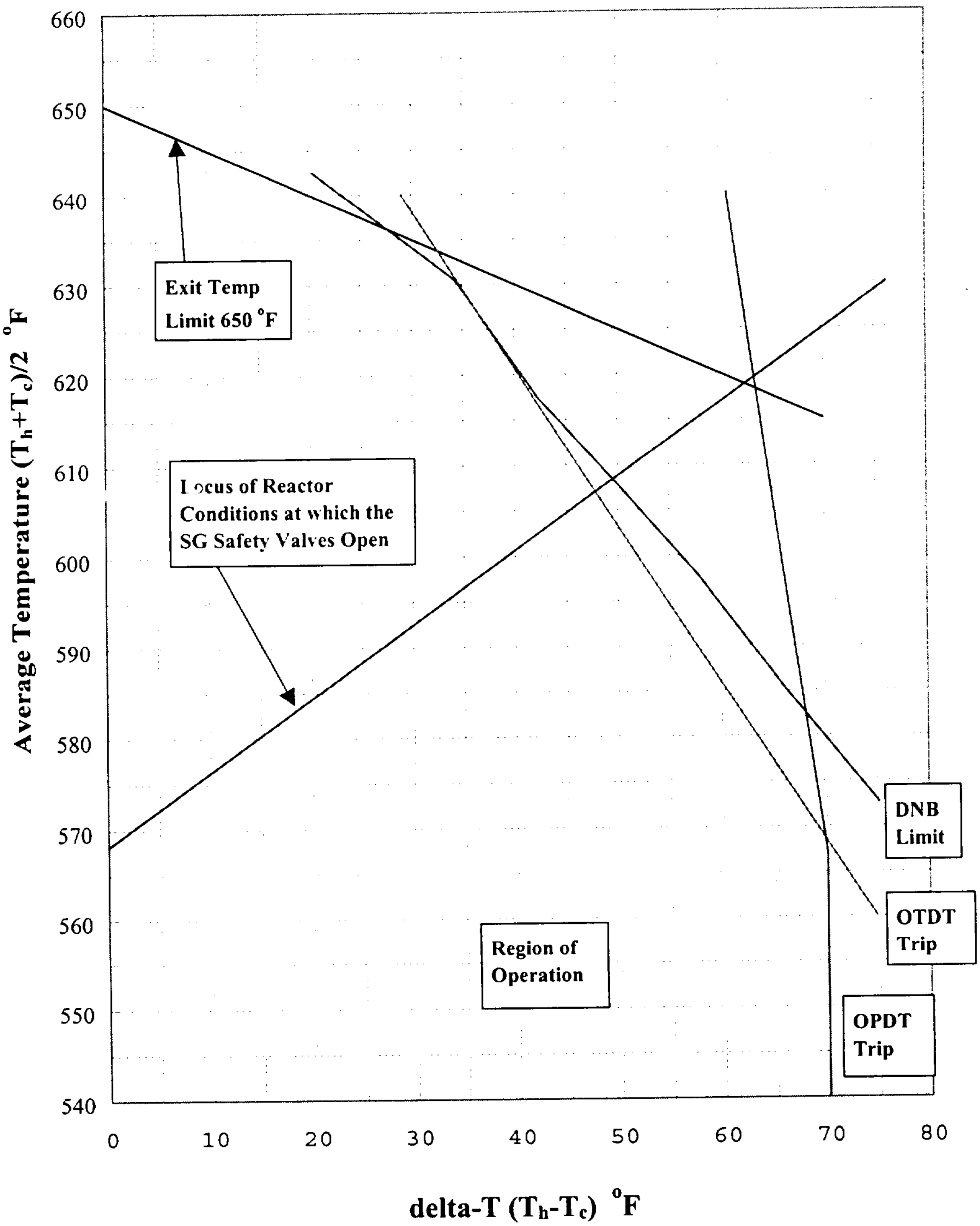

Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to AEC GDC Criterion 9, "Reactor Coolant Pressure Boundary," GDC Criterion 33, "Reactor Coolant Pressure Boundary Capability," and GDC Criterion 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

The design pressure of the RCS is 2485 psig (Ref. 2). During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there was fuel in the core. Following inception of unit operation, RCS components are pressure tested, in accordance with the requirements of ASME Code, Section XI.

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria". If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the pressurizer high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components. The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System, together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The pressurizer high pressure trip is specifically set to provide protection against overpressurization (Ref. 3). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices (Ref. 4).

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam generator power operated relief valves;

c. Steam dump system;

d. Rod control system;

e. Pressurizer level control system; or

f. Pressurizer spray valves.

BASES (continued)

SAFETY LIMITS	The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 5) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig (Ref. 2).
APPLICABILITY	SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.
SAFETY LIMIT VIOLATIONS	If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits. The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL

BASES

SAFETY LIMIT VIOLATIONS (continued)	within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 9, 33 and 34, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 4.

3. USAR, Section 7.4.

4. USAR, Section 14.4.

5. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may

BASES

LCO 3.0.2 (continued)

be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering LCO 3.0.2 ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the

BASES

LCO 3.0.2 (continued)	Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable resulting in a new LCO not met. In this case, the Completion Times of the new Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

BASES

LCO 3.0.3 (continued)

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with system operations to ensure the stability and availability of the electrical grid. The shutdown shall be initiated so that the time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met;

b. A Condition exists for which the Required Actions have now been performed; or

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then

BASES

LCO 3.0.3 (continued)

the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "Spent Fuel Storage Pool Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

BASES

LCO 3.0.4 (continued)

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering

BASES

LCO 3.0.4 (continued)

any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception tc LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective

BASES

LCO 3.0.5 (continued)

maintenance.

An example of demonstrating the OPERABILITY of the equipmen being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential

BASES

LCO 3.0.6 (continued)

confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.13, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

BASES

LCO 3.0.6 (continued)

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.4.18 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if

BASES

LCO 3.0.7 (continued)

required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 establishes an exception to LCO 3.0.2 for Technical Specification (TS) supported systems due to an inoperability of a non-Technical Specification (non-TS) support system. The specific non-TS support system for which this exception is allowed is listed in LCO 3.0.8. This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system non-TS Required Actions. The NRC has acknowledged that the system listed in LCO 3.0.8 may be inoperable for the specified Delay Time without entering the TS supported system LCO. If the non-TS support system inoperability is not corrected within the Delay Time, then the TS supported system's Conditions and Required Actions must be entered.

BASES (continued)

LCO 3.0.9

LCO 3.0.9 is provided to clarify the unit applicability of parameters or equipment designations which are specific to one unit.

In the Specifications and Bases, parentheses and footnotes may be used to identify system, component, operating parameter, setpoints, etc. specific to one unit. These are considered an integral part of the LCOs and SRs with which compliance is required for the specified unit.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the Test Exception LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event

BASES

SR 3.0.1 (continued)	may be credited as fulfilling the performances of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition. Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status. Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.
SR 3.0.2	SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval. SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2
(continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per " basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

Also, as stated in SR 3.0.2, the 25% extension does not apply to SRs with a specified Frequency of 24 months. This is to ensure performance is within equipment performance expectations. This is consistent with present industry analysis that supports refueling cycle intervals up to, but not longer than, 24 months.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

BASES (continued)

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

BASES

SR 3.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

BASES

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on

BASES

SR 3.0.4 (continued)

inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating

BASES

SR 3.0.4 (continued)

in MODE 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to AEC GDC Criteria 27 and 28 (Ref. 1), two independent reactivity control systems must be provided which are capable of holding the reactor core subcritical from any hot standby or hot operating condition. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster control assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperatures are changed to the nominal hot zero power temperature, 547°F.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

BASES

BACKGROUND (continued)

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. The primary safety analyses that rely on the SDM limits are the boron dilution and main steam line break (MSLB) analyses.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and $\leq$ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements, at end of cycle (EOC), is based on a MSLB, as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently an RCS cooldown. This results in a reduction of the reactor coolant

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As the initial RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a return to power may occur; however, fuel damage as a result of the return to power will not cause offsite doses to exceed the 10 CFR 100 limits.

The most limiting accident at beginning of cycle (BOC) is the boron dilution accident. The required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis, that is, the time available to operators to stop the dilution event. As the unit changes MODES the volume being diluted may change, i.e., if RHR is in service, as well as the critical boron concentration due to the different temperature ranges. Thus different SDMs may be required for the different MODES and dilution flow rates. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration in the RCS.

BASES

LCO (continued)

The COLR provides the shutdown margin requirements. The MSLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM requirements in the COLR. For MSLB accidents, if the LCO is violated, there is a potential to exceed 10 CFR 100, "Reactor Site Criteria," limits. For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5 the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with $k_{eff} \geq 1.0$, the SDM requirements specified in the COLR are ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components, and the probability of a design basis accident (DBA) occurring during this time is very low. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the operator should borate with the best source available for the plant conditions.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5, the SDM is verified by comparing the RCS boron concentration to a Shutdown Boron Concentration requirement curve that was generated by taking into account:

a. Required SDM;

b. Shutdown and control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the comparison.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27 and 28, issued for comment July 10, 1967, as referenced in USAR Section 1.2.
2. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to AEC GDC Criteria 27, 28, 29, and 30 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SHUTDOWN MARGIN (SDM) or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with

BASES

BACKGROUND (continued)

other variables fixed or stable (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RATED THERMAL POWER (RTP) and normal operating temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the uncertainties in the nuclear design methods are within the expected range and that the calculational models used to generate the safety analyses are adequate.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are

BASES

APPLICABLE SAFETY ANALYSES (continued)

very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions early in the cycle (≤ 60 effective full power days (EFPD)) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists early in the cycle, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed early in the cycle, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methods are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron

BASES

APPLICABILITY (continued)

Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. Verification of measured core reactivity (SR 3.1.2.1) is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If

BASES

ACTIONS

A.1 and A.2 (continued)

operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity must be verified following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The comparison is made when the core conditions such as control rod position, moderator temperature, and samarium concentration are fixed or stable. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at beginning of cycle (BOC).

SR 3.1.2.2

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made,

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.2 (continued)

considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The required Frequency of 31 effective full power days (EFPD) is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. The SR is modified by two Notes. Note 1 states that the SR is only required to be performed after 60 EFPD. Note 2 indicates that the normalization of predicted core reactivity to the measured value may take place within the first 60 EFPD after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27, 28, 29 and 30, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 14.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Isothermal Temperature Coefficient (ITC)

BASES

BACKGROUND

According to AEC GDC Criterion 8 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The moderator temperature coefficient (MTC) relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The ITC is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the ITC is the sum of the MTC and fuel temperature coefficient. The ITC is measured directly during low power PHYSICS TESTS in order to verify analytical prediction of the MTC. The units of the isothermal temperature coefficient are pcm/°F, where 1 pcm = 1E-5 $\Delta k/k$.

The reactor is designed to operate with a negative ITC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements at beginning of cycle (BOC). Reactor cores are designed so that the BOC ITC is less than zero when THERMAL POWER is at RTP. The actual value of the ITC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed

BASES

BACKGROUND (continued)

distributed poisons to yield an ITC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure that the ITC does not exceed the limits.

The limitations on ITC are provided to ensure that the value of MTC remains within the limiting conditions assumed in the USAR accident and transient analyses.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified ITC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The ITC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The USAR (Ref. 2) contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions for the cycle exposure being evaluated to ensure that the accident results are bounding.

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive (i.e., upper limit). Such accidents include the rod withdrawal transient from either zero or RTP, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include the main steam line break.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, ITC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the ITC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values will remain within the bounds of the original accident analyses during operation.

Assumptions made in safety analyses require that the ITC be less positive than a given upper bound and more positive than a given lower bound. The ITC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit usually occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the ITC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance check at BOC on ITC provides confirmation that the ITC is behaving as anticipated and will be within limits at 70% RTP, full power, and EOC so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are

BASES

LCO (continued)

established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

If the LCO limits are not met, the assumptions of the safety analysis may not be met. The core could violate criteria that prohibit a return to criticality, or the DNBR criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

APPLICABILITY

Technical Specifications place both LCO and SR values on ITC, based on the safety analysis assumptions described above.

In MODE 1, the limits on ITC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup accidents (such as the uncontrolled rod cluster control assembly withdrawal) will not violate the assumptions of the accident analysis. The lower ITC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents at EOC will not violate the assumptions of the accident analysis since ITC becomes more negative as the cycle burnup increases and the RCS boron concentration is reduced. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

ITC must be kept within the upper limit specified in LCO 3.1.3 to ensure that assumptions made in the safety analysis remain valid. The upper limit of Condition A is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO.

BASES

ACTIONS

A.1 (continued)

If the upper ITC limit is violated at BOC, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits in the future. The ITC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the ITC measurement and computing the required bank withdrawal limits.

The control rods are maintained within the administrative withdrawal limits until a subsequent calculation verifies that ITC has been restored within its limit. As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the ITC to become more negative. Using physics calculations, the time in cycle life at which the calculated ITC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1

Exceeding the EOC ITC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If it is determined during PHYSICS TESTS that the EOC ITC value will exceed the most negative ITC limit specified in the COLR, the safety analysis and core design must be re-evaluated prior to reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm to ensure that operation near the EOC remains acceptable. The 300 ppm limit is sufficient to prevent EOC operation at or below the accident analysis MTC assumptions.

Condition C has been modified by a Note that requires Required Action C.1 to be completed whenever this Condition is entered. This is necessary to ensure that the plant does not operate at conditions where the ITC would be below the most negative limit specified in the COLR.

Required Action C.1 is modified by a Note which states that LCO 3.0.4 is not applicable. This Note is provided since the requirement to re-evaluate the core design and safety analysis prior to reaching an equivalent RTP ARO boron concentration of 300 ppm is adequate action without restricting entry into MODE 1.

D.1

If the re-evaluation of the safety analysis cannot support the predicted EOC ITC lower limit, or if the Required Actions of Condition C are not completed within the associated Completion Time the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 4 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the ITC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive ITC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC ITC value for ARO will be obtained from measurements during the PHYSICS TESTS after refueling. The ARO value can be directly compared to the BOC ITC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

Measurement of the ITC at the beginning of the fuel cycle is adequate to confirm that the ITC remains within its upper limit.

SR 3.1.3.2

This SR requires measurement of ITC at BOC prior to exceeding 70% RTP after each refueling in order to confirm compliance with the 70% RTP ITC limit in the COLR. The Frequency of "Once after each refueling prior to THERMAL POWER exceeding 70% RTP" ensures the limit will be met prior to being applicable.

SR 3.1.3.3

This SR requires measurement of ITC at BOC prior to exceeding 70% RTP after each refueling in order to confirm compliance with the most negative ITC LCO. Meeting this limit prior to exceeding 70% RTP ensures that the limit will also be met at EOC.

The ITC value for EOC is derived from the ITC low power PHYSICS TESTS. The EOC value is calculated using the predicted EOC ITC from the core design report and the difference between the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.3 (continued)

measured and predicted BOC ITC. The predicted EOC value is directly compared to the most negative EOC value established in the COLR to ensure that the predicted EOC negative MTC value is within the safety analysis assumptions.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 8, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC Criteria 6, 14, 27, and 28 (Ref. 1), and 10 CFR 50.46 (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

BASES

BACKGROUND (continued)

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Both units have four control banks and two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent indications, which are the bank demand position indication (usually the group step counters) and the individual Rod Position Indication (RPI) System.

The bank demand position indication counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The bank

BASES

BACKGROUND (continued)

demand position indication is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides a highly reliable indication of rod position, but at a lower accuracy than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The RPI System is designed with an accuracy of ± 5% (approximately 12 steps) of full rod travel. There are inaccuracies arising from the normal range of coolant temperature variation from hot shutdown to full power which are compensated for by allowing ± 24 steps at the lower and upper ends of rod travel.

With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. At the lower and upper ends of rod travel with an indicated deviation of 24 steps between the group step counter and RPI, the deviation between actual rod position and the demand position could be 36 steps, or 22.5 inches.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment assure that:

a. There are no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

The safety analysis regarding static rod misalignment considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on DNBR in this case bounds the situation when a rod is misaligned from its group by 24 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA fully withdrawn (Ref. 3).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected, that the linear heat rates (LHRs) are not significantly affected, or that THERMAL POWER will be adjusted so that excessive local LHRs will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned the assumed power distribution used in the safety analysis may not be preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping.

BASES

APPLICABLE SAFETY ANALYSES (continued)	Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F_{\Delta H}^N$ to the operating limits. Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. The rod alignment requirements are satisfied when individual actual rod positions are within 24 steps of their group step counter demand position when the demand position is between 30 and 215 steps, or within 36 steps of their group step counter demand position when the demand position is $\leq$ 30 steps, or $\geq$ 215 steps. The requirement to maintain the rod alignment to within plus or minus 12 steps when the group step counter demand position is between 30 and 215 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

BASES

LCO (continued)

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are normally bottomed and the reactor is shutdown and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

BASES

ACTIONS (continued)

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

The Completion Time of 1 hour represents the time necessary for determining the unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.1.1, B.2.1.2, B.2.2, B.3, and B.4

For continued operation with a misaligned rod, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$ must be verified within limits or the high neutron

BASES

ACTIONS

B.2.1.1, B.2.1.2, B.2.2, B.3, and B.4 (continued)

flux trip setpoint must be reduced, SDM must periodically be verified within limits, the safety analyses must be re-evaluated to confirm continued operation is permissible, and, if necessary, the power level must be reduced to a level consistent with the safety analysis. Considerations in these analyses include the potential ejected rod worth and associated transient power distribution peaking factors. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod.

Verifying that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within the required limits (i.e., SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1) ensures that current operation at RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 8 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F_{\Delta H}^N$.

In lieu of determining hot channel factors ($F_Q(Z)$ and $F_{\Delta H}^N$) within the Completion Time of 8 hours, reducing the high neutron flux trip setpoint to 85% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. The Completion Time of 8 hours gives the operator sufficient time to accomplish an orderly power reduction and setpoint change without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

BASES

ACTIONS

B.2.1.1, B.2.1.2, B.2.2, B.3, and B.4 (continued)

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analyses to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in Ref. 3 that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions.

If the analyses do not support continued operation at RTP, then the power must be reduced to a level consistent with the safety analyses.

A Completion Time of 30 days is sufficient time to obtain the required input data and to perform the analysis and adjust power level.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which eliminates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

BASES

ACTIONS (continued)

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases for LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and initiate boration. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. The unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.1 is modified by a Note which direct the operators to Specification 3.1.7, "Rod Position Indication," if a rod appears to be misaligned by more than 12 steps. If the rod position indication is determined to be correct in accordance with Specification 3.1.7, then the operator must return to Specification 3.1.4 and enter the appropriate Conditions for rod misalignment.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by ≥ 10 steps will not cause radial or axial power tilts, or oscillations, to occur providing rod alignment limits are not exceeded. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.2 (continued)

control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature $\geq 500°F$ to simulate a reactor trip under actual conditions. Actual rod drop time is measured from opening of the reactor trip breaker (RTB) which is conservative with respect to beginning of decay of stationary gripper coil voltage.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 6, 14, 27, and 28, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants".

3. USAR, Section 14.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SHUTDOWN MARGIN (SDM) and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC Criteria 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution, reactivity limits, and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Some banks may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs that consists of two groups are moved in a staggered fashion, but always within one step of each other. Each reactor has four control banks and two shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks

BASES

BACKGROUND (continued)	must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.
APPLICABLE SAFETY ANALYSES	On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero

BASES

APPLICABLE SAFETY ANALYSES (continued)

power (547°F), and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits assure that:

a. There are no violations of:

   1. specified acceptable fuel design limits, or

   2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits assures that the maximum linear heat generation rate or peaking factor will be less than that used in the misaligned rod analysis. Operation at the insertion limit also assures that the maximum ejected RCCA worth will be less than the limiting value used in the ejected RCCA analysis.

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

The LCO is modified by a Note indicating that a shutdown bank may be below the insertion limit when required for performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODES 3, 4, 5, or 6, the shutdown bank insertion limit does not apply because the reactor is not producing fission power. In shutdown MODES the OPERABILITY of the shutdown rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

A.1.1, A.1.2 and A.2

With one or more shutdown banks not within insertion limits verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

BASES

ACTIONS

A.1.1, A.1.2 and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take appropriate action because the simultaneous occurrence of either an accident or transient during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 27, 28, 29, and 32, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

3. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate. The control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation ($k_{eff} \geq 1.0$) to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Some banks may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs that consists of two groups are moved in a staggered fashion, but always within one step of each other. Each reactor has four control banks and two shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

BASES

BACKGROUND (continued)

Insertion Limits

The control bank insertion limits are specified in a figure in the COLR. The control banks are required to be at or above the insertion limit lines.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. The fully withdrawn position is defined in the COLR. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature.

Overlap and Sequence

The insertion limits figure in the COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. By overlapping control bank movements, the small reactivity addition at the beginning and end of control bank travel will be compensated for; that is, the overlapping sequential movement of control banks makes the reactivity addition more uniform.

Control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B,

BASES

BACKGROUND

Overlap and Sequence (continued)

and C are at the fully withdrawn position, and control bank D is near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

General

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits assures fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant will be bounded by the safety analysis results in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other transient requiring termination by a Reactor Trip System (RTS) trip function.

BASES (continued)

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power (547°F), and to maintain the required SDM at rated no load temperature (Ref. 3). The control bank insertion limit also limits the reactivity worth of an ejected control rod.

The acceptance criteria for addressing shutdown and control bank insertion limits assure that:

a. There are no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Operation at the insertion limits assures that the maximum linear heat generation rate or peaking factor will be less than that used in the misaligned rod analysis. Operation at the insertion limit also assures that the maximum ejected RCCA worth will be less than the limiting value used in the ejected RCCA analysis.

The control bank insertion, sequence and overlap limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is limited, and ensuring adequate negative reactivity insertion is available on a trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

The LCO is modified by a Note indicating that a control bank may be below the insertion limit when required for performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would normally violate the LCO.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, and SDM. Applicability in MODE 2 with $k_{eff} < 1.0$, and in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

BASES (continued)

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 is normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either an accident or transient during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

BASES

ACTIONS (continued)

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with $k_{eff} < 1.0$, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE SURVEILLANCE

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. Prior to achieving criticality, the estimated critical position calculation appropriate for the time at which criticality is achieved shall be verified for control bank position.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. Typically, a series of ECPs are prepared in time increments applicable for a series of criticality times before and after the estimated time for achieving criticality. These ECPs account for the various factors which affect the ECP, including xenon concentration changes. The operators use the ECP applicable for the time the reactor actually achieves criticality.

SR 3.1.6.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

BASES

SURVEILLANCE (continued)

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 27, 28, 29, and 32, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors".

3. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to AEC GDC Criteria 12 and 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM). Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of

BASES

BACKGROUND (continued)

the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial positions of shutdown rods and control rods are determined by two separate and independent systems: the bank demand position indication (commonly called group step counters) and the individual Rod Position Indication (RPI) System.

The bank demand position indication counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The bank demand position indication is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides a highly reliable indication of actual control rod position, but at a lower accuracy than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The RPI System is designed with an accuracy of ± 5 % (approximately 12 steps) of full rod travel. There are inaccuracies arising from the normal range of coolant temperature variation from hot shutdown to full power which are compensated for by allowing ± 24 steps at the lower and upper ends of rod travel. With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. At the lower and upper ends of rod travel with an indicated deviation of 24 steps between the group counter and RPI, the deviation between actual rod position and the demand position could be 36 steps, or 22.5 inches.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that the RPI System and bank demand position indication be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires the following:

a. The RPI System indicates within 12 steps of the group step counter demand position when the demand position is between 30 and 215 steps, or within 24 steps of their group step counter demand position when the demand position is greater than or equal to 215 steps, or less than or equal to 30 steps; and

b. Bank demand indication has been calibrated either in the fully inserted position or to the RPI System. Demand position indication may be provided by various means such as step counters, Emergency Response Computer System (ERCS), calculations using rod drive cabinet counters or Pulse to Analog counters.

BASES

LCO (continued)

The 12 step agreement limit between bank demand position indication and the RPI System when the demand position is between 30 and 215 steps indicates that the bank demand position indication is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given above, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the RPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. See LCO 3.1.1, “SHUTDOWN MARGIN (SDM),” for SDM requirements in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, “Boron Concentration,” for boron concentration requirements during MODE 6.

BASES (continued)

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable RPI and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one RPI channel per group fails, the position of the rod may still be determined indirectly by use of the moveable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. Verification may determine that the RPI is OPERABLE and the rod is misaligned, then the Conditions of LCO 3.1.4, "Rod Group Alignment Limits" must be entered.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

BASES

ACTIONS (continued)

B.1, B.2, B.3, and B.4

When more than one RPI channel per group fails, additional monitoring shall be performed to assure that the reactor remains in a safe condition. The demand position from the group step counters associated with the rods with inoperable position indicators shall be monitored and recorded on an hourly basis. This ensures a periodic assessment of rod position to determine if rod movement in excess of 24 steps has occurred since the last determination of rod position. If rod movement in excess of 24 steps has occurred since the last determination of rod position, the Required Action of B.3 is required.

The reactor coolant system average temperature shall be monitored and recorded on an hourly basis. Monitoring and recording of the reactor coolant system average temperature may provide early detection of mispositioned or dropped rods.

When one or more rods have been moved in excess of 24 steps in one direction, since the position was last determined, action is initiated sooner to begin verifying that these rods are still properly positioned relative to their group positions. The 4 hour allowance for completion of this action allows adequate time to complete the rod position verification using the moveable incore detectors.

The position of rods with inoperable RPIs will also continue to be verified indirectly using the moveable incore detectors every 8 hours in accordance with Required Action A.1. Using the moveable incore detectors provides further assurance that the rods have not moved.

Based on experience, normal power operation does not require excessive movement of banks. Therefore, the actions specified in this condition are adequate for continued full plant operation for up to 24 hours since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour allowed out of service time also

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

provides sufficient time to troubleshoot and restore the RPI System to operation following a component failure in the system, while avoiding the challenges associated with a plant shutdown.

C.1.1 and C.1.2

Demand position indication is provided by any of the following means: step counters; ERCS; calculations using rod drive cabinet counters and Pulse to Analog counters. With all indication for one demand position per bank inoperable, the rod positions can be determined by the RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the rod position indication of the most withdrawn rod and the rod position indication of the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This ensures that the most withdrawn and least withdrawn rod are no more than 24 steps apart (including instrument uncertainty) which bounds the accident analysis assumptions. This verification can be an examination of logs, administrative controls, or other information that shows that all RPIs in the affected bank are OPERABLE.

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

BASES

ACTIONS (continued)

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the RPI agrees with the demand position within 12 steps (between 30 and 215 steps) or within 24 steps (when ≤ 30 steps or ≥ 215 steps) ensures that the RPI is operating correctly.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 12 and 13, issued for commen July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions-MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B, requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant.

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 1).

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 1) in MODE 2 are listed below:

a. Critical Boron Concentration-Control Rods Withdrawn;

b. Critical Boron Concentration-Control Rods Inserted;

c. Control Rod Worth; and

d. Isothermal Temperature Coefficient (ITC).

Low power PHYSICS TESTS may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration-Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, bank D is at or near its fully withdrawn position. HZP is where the core is critical ($k_{eff} = 1.0$), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test could violate LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)."

b. The Critical Boron Concentration-Control Rods Inserted Test measures the critical boron concentration at HZP, with the highest worth rod bank fully inserted into the core. This test is used to give an indication of the boron reactivity coefficient.

BASES

BACKGROUND (continued)

With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted gives an indication of the Boron Reactivity Coefficient compared to the measured bank worth. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits"; LCO 3.1.5, "Shutdown Bank Insertion Limit"; or LCO 3.1.6, "Control Bank Insertion Limits."

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Dilution Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Dilution Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the

BASES

BACKGROUND (continued)

selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP using the Slope Method. The Slope Method varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The ITC at beginning of cycle (BOC), 70% RTP and at end of cycle (EOC) is determined from the ITC measured in this test. This test satisfies the requirements of SR 3.1.3.1, SR 3.1.3.2 and SR 3.1.3.3. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The above mentioned PHYSICS TESTS may require the operating control or process variables to deviate from their LCO limitations.

The USAR defines requirements for initial testing of the facility, including PHYSICS TESTS. USAR Appendix J summarizes the initial plant startup, zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in Reference 1. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. The requirements specified in the following LCOs may be suspended for PHYSICS TESTING:

BASES

APPLICABLE SAFETY ANALYSES (continued)	LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)"; LCO 3.1.4, "Rod Group Alignment Limits"; LCO 3.1.5, "Shutdown Bank Insertion Limits"; LCO 3.1.6, "Control Bank Insertion Limits"; and LCO 3.4.2, "RCS Minimum Temperature for Criticality". When these LCOs are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 535°F, and SDM is within the limits provided in the COLR. The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.
LCO	This LCO allows the reactor parameters of ITC and minimum temperature for criticality to be outside their specified limits to conduct PHYSICS TESTS in MODE 2, to verify certain core physics parameters. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from "4" to "3". Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

BASES

LCO (continued)

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, 7, and 16.e, may be reduced to "3" required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 535°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour.

BASES

ACTIONS (continued)

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 535°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 535°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If Required Action C.1 cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1 (continued)

TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 535°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

Prior to achieving criticality, the SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.4 (continued)

b. Control and shutdown bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration.

After achieving criticality, this SR is met by determining the reactivity insertion available from tripping the shutdown and control banks.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. ANSI/ANS-19.6.1-1985, "Reload Startup Physics Tests for Pressurized Water Reactors."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the values of $F_Q(Z)$ which are present during non-equilibrium situations such as load following or power ascension.

BASES

BACKGROUND (continued)

To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions. Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Large Break LOCA (LBLOCA) analysis is the analysis that determines the LCO limit for $F_Q(Z)$. The $F_Q(Z)$ assumed in the Safety Analysis for other postulated accidents is either equal to or greater than that assumed in the LBLOCA analysis. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} \; K(Z) \quad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} \; K(Z) \quad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and is based on the Small Break LOCA analysis, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For Constant Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

BASES

LCO (continued)

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results a measured value ($F_Q^M(Z)$) of $F_Q(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z)*(1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances (1.03) multiplied by a factor associated with the flux map measurement uncertainty (1.05) (Ref. 3).

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ V(Z)$$

where V(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. V(Z) is included in the COLR. The $F_Q^W(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO precludes core power distributions that could violate the assumptions in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

BASES

LCO (continued)

Violating the LCO limits for $F_Q(Z)$ may result in unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq$ 1% for each 1% by which $F^C_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux-High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F^C_Q(Z)$ and would require Power Range Neutron Flux-High trip setpoint reductions within 72 hours of the $F^C_Q(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux-High trip setpoints. Decreases in $F^C_Q(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux-High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by $\geq$ 1% for each 1% by which $F^C_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F^C_Q(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required

BASES

ACTIONS

A.4 (continued)

Action A.1, ensures that core conditions during operation at higher power levels, and future operations, are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the THERMAL POWER by $\geq 1\%$ RTP for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded.

BASES

ACTIONS (continued)

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq$ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

B.3

Reduction in the Overpower ΔT trip setpoints by $\geq$ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1, ensures that core conditions during operation at higher power levels, and future operation, are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even

BASES

ACTIONS

B.4 (continued)

when Condition B is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.4 or B.1 through B.4 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of $F_Q^C(Z)$ before exceeding 75% RTP. This ensures

BASES

SURVEILLANCE REQUIREMENTS (continued)

that some determination of $F_Q(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)*(1.0815)$ (Ref. 3). $F_Q^C(Z)$ is then compared to its specified limits. The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met during the power ascension following a refueling, including when RTP is achieved, because peaking factors generally decrease as power level is increased.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F^C_Q(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^C_Q(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Frequency of 31 effective full power days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated during the nuclear design process by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called V(Z). Multiplying the measured total peaking factor, $F^C_Q(Z)$, by V(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F^W_Q(Z)$.

The limit with which $F^W_Q(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

The V(Z) curve is provided in the COLR for discrete core elevations. Flux map data are taken for 61 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 10% inclusive; and

b. Upper core region, from 90 to 100% inclusive.

The top and bottom 10% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)}\right]$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by an appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

During the power ascension following a refueling outage, startup physics testing program controls ensure that the $F_Q(Z)$ will not exceed the values assumed in the safety analysis. These controls include flux mapping, ramp rate restrictions, and restrictions on RCCA motion. They provide the necessary controls to precondition the fuel and ensure that the reactor power may be safely increased to equilibrium conditions at or near RTP, at which time $F_Q^W(Z)$ and AFD target band are determined. Performing the Surveillance within 12 hours after achieving equilibrium conditions after each refueling after THERMAL POWER exceeds 75% RTP, ensures that the $F_Q(Z)$ limit is met when the unit is released for normal operations.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^W(Z)$, another evaluation of this factor is reuquired 12 hours after achieving equilibrium condition at this higher power level (to ensure that $F_Q^W(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

BASES (continued)

REFERENCE

1. USAR, Section 14.
2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.
3. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPD). However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency (referred to as Condition II events). The departure from nucleate boiling (DNB) design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to a value greater than the criterion listed in Reference 1. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Controlling $F^N_{\Delta H}$ precludes core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition during Condition II transients (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

For transients that may be DNB limited, the THERMAL POWER, Reactor Coolant System flow, temperature, pressure and $F^N_{\Delta H}$ are the core parameters of most importance. Except for Static Rod Cluster Control Assembly (RCCA) Misalignment and Dropped Rod events, the limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any Condition II transients. The analyses for Static RCCA Misalignment and Dropped Rod events ensure the DNB design basis is met by assuming a conservatively large value for $F^N_{\Delta H}$. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion listed in Reference 1. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all Condition II transients, except Static RCCA Misalignment and Dropped Rod events, that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 1).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition I events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses as described in the Applicable Safety Analyses section above.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a factor specified in the COLR for every 1% RTP reduction in THERMAL POWER.

BASES (continued)

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1 and A.3

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1 and reduce the Power Range Neutron Flux -High trip setpoint to ≤ 55% RTP in accordance with Required Action A.3. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1 provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.3 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

BASES

ACTIONS (continued)

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.4

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

Condition A is modified by a Note that requires that Required Actions A.2 and A.4 must be completed whenever Condition A is entered.

BASES

ACTIONS (continued)

B.1

When Required Actions A.1 through A.4 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

BASES (continued)

REFERENCES

1. USAR Section 14.
2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions in conjunction with verifying $F_Q^W(Z)$ in accordance with SR 3.2.1.2. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 190 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

The AFD is logged manually or monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The frequency of monitoring the AFD by the unit computer is once per

BASES

BACKGROUND (continued)

minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is ≥ 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC and Transient Power Distribution methodologies (Refs. 1 and 2) entail:

a. Establishing an envelope of allowed power shapes and power densities;

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The Transient Power Distribution methodology (Ref. 2) determines a function, (V(Z)), that when applied to equilibrium $F_Q^C(Z)$ values will bound $F_Q^C(Z)$ values that could be measured at non-equilibrium conditions. This remains valid provided that the AFD is maintained within the target flux band around a target flux difference that was determined in conjunction with determining the equilibrium $F_Q^W(Z)$.

The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition II, III, and IV events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the loss of coolant accident. The most significant Condition III event is the loss of RCS flow accident. The most significant Condition II events are uncontrolled bank withdrawal at power and Rod Cluster Control Assembly (RCCA) misalignment.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors. Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labaeled as Δ flux or % ΔI.

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup and target flux difference.

With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its limits.

The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band.

BASES

LCO (continued)

Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time when THERMAL POWER ≥ 50% RTP. The cumulative penalty time is the sum of penalty times from LCO Notes 2 and 3.

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

For surveillance of the power range channels performed according to SR 3.3.1.6, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some

BASES

LCO (continued)	deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.
APPLICABILITY	AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3). Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions. At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events. Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.
ACTIONS	A.1 With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

BASES

ACTIONS (continued)

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

BASES

ACTIONS

C.1 (continued)

Condition C is modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered.

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

Reducing the power level to <15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 (continued)

the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.2

This Surveillance requires that the target flux difference be determined and updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup.

The target flux difference is determined by averaging the indicated AFD from all OPERABLE excore channels.

To ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during non-equilibrium state conditions, the Transient Power Distribution methodology, i.e. V(Z), (Ref. 2) requires SR 3.2.1.2 to be performed in conjunction with this SR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

Following a refueling outage, SR 3.2.1.2, and thus SR 3.2.3.2, are not required to be performed until equilibrium conditions are achieved. Since it may be desirable to provide the operators with some guidance for AFD control during the power ascension, a target flux difference may be posted based on design predictions.

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

REFERENCES

1. XN-NF-77-57, supplement 1(A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981.

2. Transient Power Distribution, NSPNAD-93003-A.

3. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure the assumptions used in the safety analysis remain valid by preventing an undetected change in the gross radial power distribution.

In MODE 1, the QPTR must be maintained within limits to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the assumptions in the safety analysis are possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to

BASES

APPLICABILITY (continued)

require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^{N}_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in the QPTR would require power reductions within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

BASES

ACTIONS (continued)

A.3

The peaking factor $F_Q(Z)$ (as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$) and $F_{\Delta H}^N$ are of primary importance in ensuring that them power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ for changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern

BASES

ACTIONS

A.4 (continued)

exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

BASES

ACTIONS (continued)

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable,

BASES

ACTIONS

B.1 (continued)

based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 85% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of a core power tilt that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channel inputs are inoperable and the THERMAL POWER is > 85% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that the QPTR remains within its limits.

For purposes of monitoring changes in radial core power distribution when one power range channel is inoperable, at least 2 moveable incore detectors or 4 thermocouples per quadrant may be used to calculate an incore core power tilt. This incore core power tilt may be used, instead of the excore detectors, to confirm that the QPTR is within the limits by comparing it to previous flux maps.

REFERENCES

1. USAR, Section 14.
2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

AEC GDC Criterion 14, "Core Protection Systems" (Ref. 1), requires that core protection systems, together with associated equipment, shall be designed to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits. The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting Safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical specifications are required by 10CFR50.36 to contain LSSS defined by the regulation as "... settings for automatic protective devices ... so chosen that automatic protective action will correct the abnormal situation before a safety limit (SL) is exceeded." The analytical limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the analytical limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protective devices must be chosen to be more conservative than the analytical limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The trip setpoint is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable

BASES

BACKGROUND (continued)

reaching the analytical limit and thus ensuring that the SL would not be exceeded. As such, the trip setpoint accounts for uncertainties in setting the device (e.g., calibration), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the trip setpoint plays an important role in ensuring that SLs are not exceeded. As such, the trip setpoint meets the definition of an LSSS (Ref. 2) and could be used to meet the requirement that they be contained in the technical specifications.

Technical specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in technical specifications as "... being capable of performing its safety function(s)." For automatic protective devices, the required safety function is to ensure that a SL is not exceeded and therefore the LSSS as defined by 10CFR50.36 is the same as the OPERABILITY limit for these devices. However, use of the trip setpoint to define OPERABILITY in technical specifications and its corresponding designation as the LSSS required by 10CFR50.36 would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protective device setting during a surveillance. This would result in technical specification compliance problems, as well as reports and corrective actions required by the rule which are necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the trip setpoint due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the trip setpoint and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety

BASES

BACKGROUND (continued)

function and the only corrective action required would be to reset the device to the trip setpoint to account for further drift during the next surveillance interval.

Use of the trip setpoint to define "as-found" OPERABILITY and its designation as the LSSS under the expected circumstances described above would result in actions required by both the rule and technical specifications that are clearly not warranted. However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value needs to be specified in the technical specifications in order to define OPERABILITY of the devices and is designated as the Allowable Value which, as stated above, is the same as the LSSS.

The Allowable Value specified in Table 3.3.1-1 serves as the LSSS such that a channel is OPERABLE if the actual setting is found not to exceed the Allowable Value during the CHANNEL OPERATIONAL TEST (COT). As such, the Allowable Value differs from the trip setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a safety limit is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable from a technical specification perspective. This requires corrective action including those actions required by 10CFR50.36 when automatic protective devices do not function as required.

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BACKGROUND (continued)

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits (Ref. 3) are:

1. The departure from nucleate boiling ratio (DNBR) shall be maintained to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. One acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into interconnected portions as described in the USAR (Ref. 4), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;
2. Reactor Protection Analog System and Nuclear Instrumentation System (NIS), arranged in protection channel sets: provides signal conditioning, bistable setpoint comparison, bistable electrical signal output to protection system relay logic, and control board/control room/miscellaneous indications;

BASES

BACKGROUND (continued)

3. Reactor Protection Relay Logic System, including channelized input and logic: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable outputs from the analog protection system and NIS; and

4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the trip setpoint and Allowable Values. The OPERABILITY of each transmitter or sensor is determined by either "as-found" calibration data evaluated during the CHANNEL CALIBRATION or by qualitative assessment of field transmitter or sensor as related to the channel behavior observed during performance of the CHANNEL CHECK.

Reactor Protection Analog and NI Systems

Generally, two to four channels of instrumentation are used for the signal processing of unit parameters measured by the field instruments. The instrument channels provide signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints that are based on safety analyses (Ref. 3). If the measured value of a unit parameter exceeds the predetermined setpoint, an

BASES

BACKGROUND

Reactor Protection Analog and NI Systems (continued)

output from a bistable actuates logic input relays. Channel separation is described in Reference 4. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the protection logic, main control board indication, and the plant computer. In addition, some provide input to one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function will still operate with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function will still operate with a one-out-of-two logic.

If a parameter is used for input to the protection logic and a control function, the circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. These requirements are described in IEEE-279-1971. Where necessary to provide the required reliability and redundancy, four channels with a two-out-of-four logic, or median signal selection and signal validation are provided. The actual number of channels required for each unit parameter is specified in Reference 4. Again, a single failure will neither cause nor prevent the protection function actuation.

Allowable Values and RTS Setpoints

The trip setpoints used in the bistables are based on the analytical limits from Reference 3. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration

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BACKGROUND

Allowable Values and RTS Setpoints (continued)

tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49, the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservative with respect to the analytical limits. A detailed description of the methodology used to calculate the Allowable Values and trip setpoints, including their explicit uncertainties, is provided in the plant specific setpoint methodology study (Ref. 5) which incorporates all of the known uncertainties applicable to each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint and corresponding Allowable Value. The trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value (LSSS) to account for measurement errors detectable by the COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

The trip setpoint is the value at which the bistable is set and is the expected value to be achieved during calibration. The trip setpoint value ensures the LSSS and the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable is considered to be properly adjusted when the "as-left" setpoint value is within the band for CHANNEL CALIBRATION uncertainty allowance (i.e., ± rack calibration + bistable setting uncertainties).

Trip setpoints consistent with the requirements of the Allowable Value ensure that SLs and DNBR limits are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed).

Each required instrument channel can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance

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BACKGROUND

Allowable Values and RTS Setpoints (continued)

requirements of Reference 5. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The instrumentation for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

Reactor Protection Relay Logic System

The relay logic equipment uses outputs from the analog and NI bistables. To meet the redundancy requirements, two trains of relay logic, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own set of cabinets for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to provide a reactor trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The bistable outputs are combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. During normal operation the output from

BASES

BACKGROUND

Reactor Trip Switchgear (continued)

the relay logic is a contact signal that energizes the undervoltage coils in the RTBs, and bypass breakers if in use. When the required logic matrix combination is completed, the relay logic output contacts open, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the relay logic. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself to open the RTBs, thus providing a diverse trip mechanism.

The logic matrix Functions are described in Reference 4. In addition to the reactor trip or ESF, Reference 4 also identifies the various "permissive interlocks" that are associated with unit conditions. Each train has built in test features that allow testing of the logic matrix Functions while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The RTS functions to maintain the SLs and DNBR limits during AOOs as identified in Reference 3 and mitigates the consequences of DBAs in all MODES in which the Rod Control System is capable of rod withdrawal or one or more rods not fully inserted.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the safety analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

conditions that do not require dynamic transient analysis. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. A channel is OPERABLE with an actual setpoint value outside its calibration tolerance provided the actual setpoint "as-found" value does not exceed its associated Allowable Value and provided the setpoint "as-left" value is adjusted to a value within the "as-left" calibration tolerance band. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of two, three, or four channels in each instrumentation Function, two channels of Manual Reactor Trip and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when RTS channels are also used as control system inputs and there is a possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint.

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breakers in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, Manual Reactor Trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the Manual Reactor Trip Function must also be OPERABLE if one or more shutdown rods or control rods are withdrawn or the Rod Control System is capable of withdrawing the shutdown rods or control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, Manual Reactor Trip does not have to be OPERABLE if the Rod Control System is not capable of withdrawing the shutdown rods or control rods and if all rods are fully inserted. If the rods cannot be withdrawn from the core, or all of the rods are inserted there is no need to be able to trip the reactor. In MODE 6, the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the Manual Reactor Trip Function is not required.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Reactor Trip System Functions (continued)

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the reactor control system. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor. While performing PHYSICS TESTS in accordance with LCO 3.1.8, the number of required channels may be reduced to three.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires all four of the Power Range Neutron Flux-High channels to be OPERABLE.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range channels cannot indicate neutron levels in this range. In these MODES, the Power Range Neutron Flux-High does not have to be OPERABLE because the reactor is shut down

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Power Range Neutron Flux-High (continued)

and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux P-10 setpoint, and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two-out-of-four power range channels are greater than the P-10 setpoint. This Function is automatically unblocked when three-out-of-four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range channels cannot indicate neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

a. Power Range Neutron Flux-High Positive Rate

The Power Range Neutron Flux-High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA rod drive housing rupture and the accompanying ejection of the RCCA and uncontrolled RCCA withdrawal at power. This Function compliments the Power Range Neutron Flux-High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux-High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident or uncontrolled RCCA withdrawal at power, the Power Range Neutron Flux-High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range channels cannot indicate neutron levels present in this mode.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Power Range Neutron Flux Rate (continued)

b. Power Range Neutron Flux-High Negative Rate

The Power Range Neutron Flux-High Negative Rate trip Function ensures that protection is provided for rod drop events. At high power levels, a rod drop event could cause local flux peaking that would result in an unconservative DNBR. Operation of this Function is not required for certain single rod drop events. Safety analysis results show that DNBR will be greater than the limit for those dropped rods that do not actuate this function.

The LCO requires all four Power Range Neutron Flux-High Negative Rate channels to be OPERABLE.

In MODE 1 or 2, when there is potential for a rod drop event to occur, thc Power Range Neutron Flux-High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range channels cannot indicate neutron levels present in this MODE.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function provides redundant protection to the Power Range Neutron Flux-Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Intermediate Range Neutron Flux (continued)

do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal event during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux-High Setpoint trip and the Power Range Neutron Flux-High Positive Rate trip provide core protection for a rod withdrawal event. In MODE 2 below the P-6 setpoint, and in MODE 3, 4, or 5, the Source Range Neutron Flux Trip provides the core protection for reactivity events. The core also has the required SDM to mitigate the consequences of a positive reactivity addition event. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range channels cannot indicate neutron levels present in this MODE.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition. This trip Function provides redundant protection to the Power Range Neutron Flux-Low trip Function. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5 when rods are capable of withdrawal or one or more rods are not fully inserted. Therefore, the functional capability at the specified trip setpoint is assumed to be available.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events.

In MODE 2 when below the P-6 setpoint and in MODES 3, 4, and 5 when there is a potential for an uncontrolled RCCA bank rod withdrawal event the Source Range Neutron Flux trip must be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low trip will provide core protection for reactivity events. Above the P-6 setpoint, the NIS source range detectors are de-energized .

In MODES 3, 4, and 5 with all rods fully inserted and the Rod Control System not capable of rod withdrawal, and in MODE 6, the outputs from the Function to RTS logic are not required to be OPERABLE.

The requirements for the NIS source range detectors to monitor core neutron levels and provide indication of reactivity changes

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

that may occur as a result of events like a boron dilution are addressed in LCO 3.9.3, "Nuclear Instrumentation," for MODE 6.

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. The inputs to the Overtemperature ΔT trip include pressurizer pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the trip setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure – the trip setpoint is varied to correct for changes in system pressure; and
- axial power distribution – $f(\Delta I)$, the trip setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range channels. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range channels, the trip setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature ΔT trip Function is calculated for each channel as described in Note 1 of Table 3.3.1-1. A trip occurs if Overtemperature ΔT is indicated in two channels. Since the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the trip setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions. While performing PHYSICS TESTS in accordance with LCO 3.1.8, the number of required channels may be reduced to three.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the trip setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- rate of change of reactor coolant average temperature – including dynamic compensation for the delays between the core and the temperature measurement system; and
- axial power distribution – f(ΔI), the trip setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the trip setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower ΔT trip Function is calculated for each channel as per Note 2 of Table 3.3.1-1. A trip occurs if Overpower ΔT is indicated in two channels. Since the temperature signals are used for other control functions, the actuation logic must be

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions. While performing PHYSICS TESTS in accordance with LCO 3.1.8, the number of required channels may be reduced to three.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and-Low trips and the Overtemperature ΔT trip.

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Pressurizer Pressure-Low (continued)

The LCO requires four channels of Pressurizer Pressure -Low to be OPERABLE. Since the pressurizer pressure channels are also used to provide input to the pressurizer pressure control system, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure- Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range or turbine impulse pressure). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, the AOO's meet their DNB criteria without requiring this trip function.

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires three channels of the Pressurizer Pressure-High to be OPERABLE. Although the pressurizer pressure channels are also input to pressure control, an input failure to the control system can not cause a transient that would require actuation of this protection Function. Three OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Pressurizer Pressure-High (continued)

The Pressurizer Pressure-High Allowable Value is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure-High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level-High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available,

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Pressurizer Water Level-High (continued)

pressure overshoot due to level channel failure cannot cause the safety valve to lift before the reactor high pressure trip.

In MODE 1, when there is a potential for transients such as a load rejection causing overfill of the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, a loss of flow in either RCS loop will actuate a reactor trip. Above the P-7 setpoint, a loss of flow in both RCS loops will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input. This flow is a percent of normal indicated loop flow as measured at loop elbow tap.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE in MODE 1 above P-7 or P-8.

In MODE 1 above the P-7 or P-8 setpoints, a loss of flow in an RCS loop could result in DNB conditions in the core. Below the P-7 and P-8 setpoints, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Loss of Reactor Coolant Pump (RCP)

Loss of RCP trip Function operates on two sets of auxiliary contacts, with one set on each RCP breaker. This Function anticipates the Reactor Coolant Flow-Low trips to avoid RCS heatup that would occur before the low flow trip actuates.

a. Reactor Coolant Pump Breaker Open

The RCP Breaker Open trip Function provides protection against violating the DNBR limit due to a loss of flow in one or both RCS loop(s). The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. If both RCP breakers are open above the P-7 setpoint, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low trip setpoint is reached.

The LCO requires one RCP Breaker Open channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this trip Function because the RCS Flow- Low trip alone provides sufficient protection of the DNBR limit for loss of flow events. The RCP Breaker Open trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using position switches. Therefore, the Function has no adjustable trip setpoint with which to associate an allowable value.

In MODE 1 above the P-7 or P-8 setpoints, when a loss of low in either or both RCS loop(s) could result in DNB conditions in the core, the RCP Breaker Open trip must be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

a. Reactor Coolant Pump Breaker Open (continued)

Below the P-7 and P-8 setpoints, RCP Breaker Open reactor trips are automatically blocked since the analyses demonstrate AOOs meet their DNB criteria without requiring this trip function at this low power level. Above the P-7 or P-8 setpoints, the RCP Breaker Open reactor trips are automatically enabled.

b. Underfrequency 4 kV Buses 11 and 12 (21 and 22)

The Underfrequency 4 kV Buses 11 and 12 (21 and 22) breaker trip Function provides protection against violating the DNBR limit due to a loss of flow in both RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. A loss of frequency detected on both RCP buses will initiate a trip of both RCP breakers. This trip will generate a reactor trip before the Reactor Coolant Flow-Low trip setpoint is reached. Time delays are incorporated into the Underfrequency 4 kV Buses 11 and 12 (21 and 22) channels to prevent RCP breaker trips due to momentary electrical power transients.

The LCO requires two underfrequency channels per bus to be OPERABLE.

In MODE 1 above the P-7 or P-8 setpoints, the Underfrequency 4 kV Buses 11 and 12 (21 and 22) trip Function must be OPERABLE. Below the P-7 and P-8 setpoints, reactor trips on RCP Breaker Open are automatically blocked since the AOOs meet their DNB criteria without requiring this trip Function at this low

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

12. Undervoltage on 4 kV Buses 11 and 12 (21 and 22)

power level. Above the P-7 or P-8 setpoints, the reactor trips on RCP Breaker Open are automatically enabled.

The Undervoltage on 4 kV Buses 11 and 12 (21 and 22) Function provides protection against violating the DNBR limit due to a loss of flow in both RCS loops. The voltage to each loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on both RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low trip setpoint is reached. Time delays are incorporated into the undervoltage channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two undervoltage channels per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the undervoltage trip must be OPERABLE. Below the P-7 setpoint, reactor trips on undervoltage are automatically blocked since analyses demonstrate AOOs meet their DNB criteria without requiring this trip Function at this low power level. Above the P-7 setpoint, the reactor trip on undervoltage in both RCS loops is automatically enabled. This Function uses the same relays as the ESFAS Function 6.d, "Undervoltage on 4 kV Buses 11 and 12 (21 and 22)" start of the turbine driven auxiliary feedwater pump.

13. Steam Generator (SG) Water Level-Low Low

The SG Water Level-Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the Auxiliary Feedwater (AFW) System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

13. Steam Generator (SG) Water Level-Low Low (continued)

water. A narrow range low low level in either SG indicates a potential loss of heat sink for the reactor. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires three channels of SG Water Level-Low Low per SG to be OPERABLE. The level channels provide input to the SG level control system. However, median signal selection ensures that the failure of a single channel will not result in a low level which may require the protection function actuation. Therefore, only three channels per SG are required to be OPERABLE.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level-Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). Generally the MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level-Low Low Function does not have to be OPERABLE because the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System or by the Residual Heat Removal (RHR) System in MODE 3, 4, 5, or 6.

14. Turbine Trip

a. Turbine Trip-Low Autostop Oil Pressure

The Turbine Trip-Low Autostop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

a. Turbine Trip-Low Autostop Oil Pressure (continued)

Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint will not actuate a reactor trip. Three pressure switches monitor the autostop oil pressure in the turbine electrohydraulic control system. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip-Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip-Low Autostop Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Turbine Trip-Turbine Stop Valve Closure (continued)

operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Autostop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch channel that inputs to the RTS relay logic. If the limit switches indicate that both stop valves are closed, a reactor trip is initiated.

The allowable value for this Function is set to assure channel trip occurs when the associated stop valve is closed.

The LCO requires two Turbine Trip-Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. Both channels to a train of relay logic must trip to cause reactor trip.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation relay logic will initiate a reactor trip upon any signal that initiates SI. This is not a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion,

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (continued)

except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present. Allowable Values are not applicable to this Function. The SI Input is provided by relays in the ESFAS. Therefore, there is no measurement signal with which to associate an Allowable Value.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed.

Each interlock Function consists of the following circuitry:

- The bistables that provide the applicable process parameter input;
- Logic input relays and contact matrix; and

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

16. Reactor Trip System Interlocks (continued)

- Permissive (P) relays that provide the interlock to the appropriate trip logic.

The interlock Functions are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and

- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Intermediate Range Neutron Flux, P-6 (continued)

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either Power Range Neutron Flux or Turbine Impulse Pressure. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure-Low;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (both loops);
- RCPs Breaker Open (both loops); and
- Undervoltage 4 kV Buses 11 and 12 (21 and 22).

These reactor trips are only required when operating above the P-7 setpoint. The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure- Low;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (both loops);
- RCP Breaker Position (both loops); and
- Undervoltage 4 kV Buses 11 and 12 (21 and 22).

The associated trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below the P-7 setpoint, which is in MODE 1.

1. Power Range Neutron Flux, P-7

Power Range Neutron Flux, P-7 is actuated by two-out-of-four NIS power range channels. The LCO requirement for this Function ensures that this input to the P-7 interlock is available.

The LCO requires four channels of Power Range Neutron Flux, P-7 to be OPERABLE in MODE 1.

OPERABILITY in MODE 1 ensures the Function is available to perform its increasing power Functions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Turbine Impulse Pressure, P-7

The Turbine Impulse Pressure, P-7 is actuated by one-out-of-two pressure channels. The LCO requirement for this Function ensures that this input to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure, P-7 to be OPERABLE in MODE 1. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated by two-out-of-four NIS power range channels. The P-8 interlock automatically enables the Reactor Coolant Flow-Low (single loop) and RCP Breaker Open (single loop) reactor trips on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than the P-8 setpoint. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated by two-out-of-four NIS power range channels. The LCO requirement for this Function ensures that the Turbine Trip-Low Autostop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip could cause a load rejection beyond the capacity of the steam dump and Rod Control Systems. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the steam dump and Rod Control Systems, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the steam dump system.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated by two-out-of-four NIS power range channels. If power level falls below the setpoint on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

e. Power Range Neutron Flux, P-10 (continued)

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;

- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux-Low reactor trip;

- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors; and

- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2. OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection. While performing PHYSICS TESTS in accordance with LCO 3.1.8, the number of required channels may be reduced to three.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

17. Reactor Trip Breakers

The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Rod Control System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability. The OPERABILITY requirement for the individual trip mechanisms is provided in the Function 18 below.

This trip Function must be OPERABLE in MODE 1 or 2. This ensures it is available when the reactor is critical. In MODE 3, 4, or 5, this RTS trip Function must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2. This ensures they are available when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

19. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE trains ensures that random failure of a single logic train will not prevent reactor trip.

This trip Function must be OPERABLE in MODE 1 or 2. This ensures it is available when the reactor is critical. In MODE 3, 4, or 5, this RTS trip Function must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods not fully inserted.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO

BASES

ACTIONS (continued)

Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified (e.g., on a per steamline, per loop, per SG, etc., basis), then the Condition may be entered separately for each steamline, loop, SG, etc., as applicable.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit may be outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the Reactor Protection Relay Logic for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

BASES

ACTIONS

B.1 and B.2 (continued)

If the channel cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, Action C would apply to any inoperable Manual Reactor Trip Function if the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

C.1 , C.2.1 and C.2.2

Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted:

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the RTS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, action must be initiated within the

BASES

ACTIONS

C.1 , C.2.1 and C.2.2 (continued)

same 48 hours to ensure that all rods are fully inserted and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With rods fully inserted and the Rod Control System incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

D.1.1, D.1.2, and D.2

Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux-High Function;
- Power Range Neutron Flux-Low;
- Power Range Neutron Flux-High Positive Rate;
- Power Range Neutron Flux-High Negative Rate.

The NIS power range detectors provide input to the reactor control system and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 6).

BASES

ACTIONS

D.1.1, D.1.2, and D.2 (continued)

In addition to placing the inoperable channel in the tripped condition, monitor the QPTR once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels > 85% RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

If Condition D is entered while performing PHYSICS TESTS in accordance with LCO 3.1.8, a total of two channels may be inoperable.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 6.

Required Action D.1.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if THERMAL POWER is > 85% RTP and the Power Range Neutron Flux input to QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function

BASES

ACTIONS

D.1.1, D.1.2, and D.2 (continued)

inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-High; and
- SG Water Level-Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 6.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up

BASES

ACTIONS

E.1 and E.2 (continued)

to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range channel performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase THERMAL POWER above the P-10 setpoint. These actions are consistent with the provisions of LCO 3.0.4 which stats, " This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS . . ." The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range channels perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

BASES

ACTIONS (continued)

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range channel performs the monitoring Functions.

With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

Required Action G.1 is modified by a Note to indicate that normal plant control operations that individually add limited reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

H.1

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

BASES

ACTIONS

H.1 (continued)

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is reduced and any actions that add positive reactivity to the core must be suspended immediately.

Required Action H.1 is modified by a Note to indicate that normal plant control operations that individually add limited reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, or in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

J.1 and J.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an

BASES

ACTIONS

J.1 and J.2 (continued)

OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour are justified in Reference 6.

K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure-Low;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (single loop); and
- Reactor Coolant Flow-Low (both loops).

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 or P-8 setpoints. These Functions do not have to be OPERABLE below the P-7 and P-8 setpoints because there are no loss of flow trips below these setpoints. There is insufficient heat production to generate DNB conditions below these setpoints. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 6. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 and P-8 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

BASES

ACTIONS

K.1 and K.2 (continued)

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channels, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

L.1 and L.2

Condition L applies to the Loss of Reactor Coolant Pump Underfrequency 4 kV Buses 11 and 12 (21 and 22) and Undervoltage on 4 kV Buses 11 and 12 (21 and 22). With one or both channels inoperable on one bus, the inoperable channel(s) must be placed in trip within 6 hours. If the channel(s) cannot be restored to OPERABLE status or the channel(s) placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-7 and P-8 setpoints within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable. These trip Functions do not have to be OPERABLE below the P-7 and P-8 setpoints because analyses demonstrate AOOs meet their DNB criteria without requiring these trip functions at this low power level. The 6 hours allowed to restore the channel(s) to OPERABLE status or place in trip and the 6 additional hours allowed to reduce THERMAL POWER to below the P-7 and P-8 setpoints are justified in Reference 6.

The Required Actions have been modified by a Note that allows placing one inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

BASES

ACTIONS (continued)

M.1 and M.2

Condition M applies to the RCP Breaker Open reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains, other breaker position channels, other flow related trip Functions and the low probability of an event occurring during this interval.

If the channel cannot be restored to OPERABLE status within the 48 hours, then THERMAL POWER must be reduced below the P-7 and P-8 setpoints within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable. This Function does not have to be OPERABLE below the P-7 and P-8 setpoints because analyses demonstrate AOOs meet their DNB criteria without requiring this Trip Function at this low power level.

N.1 and N.2

Condition N applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 6 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 6 hours allowed for reducing power are justified in Reference 6.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channel(s). The 4 hour time limit is justified in Reference 6.

BASES

ACTIONS (continued)

O.1 and O.2

Condition O applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action O.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action O.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of an additional 6 hours (Required Action O.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 8 hours for surveillance testing, provided the other train is OPERABLE. A train is normally bypassed by placing the bypass breaker in service and opening the associated RTB. The RTB remains OPERABLE under these conditions so that entry into Condition P is not required while performing testing allowed by this Note.

P.1 and P.2

Condition P applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one RTB train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 7 hour Completion Times are equal to the

BASES

ACTIONS

P.1 and P.2 (continued)

time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 results in Action C entry while RTB(s) are inoperable.

The Required Actions have been modified by two Notes. Note 1 allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 4 hours for maintenance on undervoltage or shunt trip mechanisms if the other train is OPERABLE.

Q.1 and Q.2

Condition Q applies to the P-6 and P-10 interlocks. With one or more channel(s) inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status ensures the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 7 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

R.1 and R.2

Condition R applies to the P-7, P-8, and P-9 interlocks. With one or more channel(s) inoperable for one-out-of-two or two-out-of-four

BASES

ACTIONS

R.1 and R.2 (continued)

coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status ensures the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

S.1 and S.2

Condition S applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of an additional 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

With the unit in MODE 3, Action C would apply to any inoperable RTB Trip mechanism. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 4 hours, per Condition P.

BASES

ACTIONS

S.1 and S.2 (continued)

The Completion Time of 48 hours for Required Action S.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of reactor protection analog system supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 Effective Full Power Days (EFPD). If the absolute difference is ≥ 2%, the NIS channel is still OPERABLE, but must be readjusted.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the overtemperature and overpower ΔT Functions.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 2%.

Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 72 hours is allowed for performing the first Surveillance after reaching 15% RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Verification of the shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test. When performing this SR, manually trip the UV trip attachment remotely (i.e., from the protection system racks). A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The RTS relay logic is tested every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All logic combinations, with applicable permissives, are tested for each protection function required for the current plant MODE. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature and overpower ΔT Functions.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 75% RTP and that 24 hours is allowed for performing the first surveillance after reaching 75% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days. A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as-found" and "as-left" values must also be recorded and reviewed for consistency with the assumptions of Reference 5.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3.

The Frequency of 92 days is justified in Reference 6.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by two Notes. Note 1 requires that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. Verification that P-6 and P-10 are in their required state for existing plant conditions can also be accomplished by observation of the permissive annunciator window. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. Note 2 provides an exception from performance of this SR prior to reactor startup for the intermediate and source range instrumentation when the reactor has been shutdown less than or equal to 48 hours. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within the previous 92 days. The Frequency of "prior to reactor startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of 12 hours after reducing power below P-10 (applicable to intermediate and power range low channels) and 4 hours after reducing power below P-6 (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and twelve and four hours after reducing power below P-10 or P-6, respectively. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than twelve hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the time limit. Twelve hours and four hours are reasonable times to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 12 and 4 hours, respectively.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 6. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to undervoltage and underfrequency relays, setpoint verification is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor where applicable (e.g., the undervoltage and underfrequency relays do not have separate sensors). The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.10 (continued)

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. This Surveillance includes verification that the time constants, where applicable, are adjusted to the prescribed values. The CHANNEL CALIBRATION for the power range neutron detectors is performed in accordance with SR 3.3.1.2 and SR 3.3.1.6. The 24 month Frequency is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 24 months. This SR is modified by a Note stating that this test shall include verification of the RCS resistance temperature detector (RTD) bypass loop flow rate. This Surveillance includes verification that the time constants, where applicable, are adjusted to the prescribed values.

The Frequency is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every 24 months. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. This TADOT is performed every 24 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers. The Reactor Trip Bypass Breaker test shall include testing of the undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.14 (continued)

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This test shall verify OPERABILITY by actuation of the end device. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. This test is performed prior to exceeding P-9 interlock whenever the unit has been in MODE 3. This Surveillance is required if not performed within the previous 31 days. A Note states that verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in appropriate plant procedures. Individual component response times are typically not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

Response time test is performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.16 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 14, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation."

3. USAR, Section 14.

4. USAR, Section 7.

5. "Engineering Manual Section 3.3.4.1, Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations".

6. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.